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10CFR 50.73

November 7, 2003

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Peach Bottom Atomic Power Station (PBAPS) Units 2 & 3
Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 & 50-278

Subject: Licensee Event Report (LER) 2-03-04

This LER reports automatic scrams and other plant operational events for Units 2 and 3 that resulted from an off-site electrical grid disturbance that occurred approximately 35 miles away from the site. In accordance with NEI 99-04, the regulatory commitment contained in this correspondence is to restore compliance with the regulations. The specific methods that are planned to restore and maintain compliance are discussed in the LER. If you have any questions or require additional information, please do not hesitate to contact us.

Sincerely,

John A. Stone

John A. Stone
Plant Manager
Peach Bottom Atomic Power Station

JAS/djf/CR 175737

Attachment

cc: PSE&G, Financial Controls and Co-owner Affairs
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INPO Records Center
H. J. Miller, US NRC, Administrator, Region I
R. I. McLean, State of Maryland
C. W. Smith, US NRC, Senior Resident Inspector

CCN 03-14091

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SUMMARY OF EXELON NUCLEAR COMMITMENTS

The following table identifies commitments made in this document by Exelon Nuclear. (Any other actions discussed in the submittal represent intended or planned actions by Exelon Nuclear. They are described to the NRC for the NRC's information and are not regulatory commitments.)

Commitment	Committed Date or "Outage"
In accordance with NEI 99-04, the regulatory commitment contained in this correspondence is to restore compliance with the regulations. The specific methods that are planned to restore and maintain compliance are discussed in the LER.	In accordance with the Corrective Action Program

NRC FORM 366 (7-2001)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Peach Bottom Atomic Power Station, Units 2 (and 3)	05000 277	1 OF 9

4. TITLE

Units 2 and 3 Automatic Scrams Resulting from an Off-Site Electrical Grid Disturbance

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME: Peach Bottom 3	DOCKET NUMBER: 50-278
09	15	03	03	– 04 –	00	11	07	03	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	1
10. POWER LEVEL	100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)	50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)	50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)	73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)	73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)	OTHER Specify in Abstract below or in NRC Form 366A
	[illegible]		50.46(a)(3)(ii)		[illegible]	
	[illegible]		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)	
	[illegible]		[illegible]		50.73(a)(2)(vii)	
	[illegible]		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)	
	[illegible]	X	50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Ellen P. Anderson - Regulatory Assurance Manager	(717) 456-3588

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
E	EK	SEAL	F010	Y	X	SB	ISV	A660	Y

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

At approximately 0132 on 9/15/03, Units 2 and 3 automatically scrammed and received Primary Containment Isolations as a result of an interruption of power to the Reactor Protection System (RPS) and the Primary Containment Isolation System (PCIS) logic circuits. This interruption of power was caused by a brief loss of two of the three PBAPS off-site power sources caused by an electrical grid disturbance approximately 35 miles away from the site. The disturbance was the result of failure of off-site protective relaying during a lightning storm. The Emergency Diesel Generators (EDGs) started and provided on-site power. On Unit 3, one Safety Relief Valve (SRV) remained open after actuation. It subsequently closed when reactor pressure was reduced. At approximately 0235 hours, the E-2 EDG tripped on low jacket coolant pressure. A discretionary Unusual Event was declared by the Shift Manager as a result of the E-2 EDG trip combined with the off-site grid concerns. The High Pressure Coolant Injection and Reactor Core Isolation Cooling systems were used to provide reactor water level control. Safety Relief Valves were used for reactor pressure control. The Suppression Pool cooling system was used for containment heat removal. Normal power was restored to the on-site emergency busses and the Unusual Event was terminated at 1046 hours. The cause of the event was due to less than adequate maintenance and testing on protective relaying on the off-site electrical distribution system. Appropriate maintenance, testing and other upgrades are being pursued.

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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

Unit Conditions Prior to the Event

Unit 2 was in Mode 1 and operating at approximately 100% rated thermal power when the event occurred. Unit 3 was in Mode 1 at approximately 90% rated thermal power in end-of-cycle coast down when this event occurred. At the time of the event, there were no structures, systems or components out of service that contributed to this event. The station was in a normal electrical system line-up and there was no maintenance or testing in progress on the station electrical system.

Description of the Event

At approximately 0132 on 9/15/03, Units 2 and 3 automatically scrammed and received Primary Containment Isolations as a result of an interruption of power to the Reactor Protection System (RPS) (EIIS: JC) and the Primary Containment Isolation System (PCIS) (EIIS: JM) logic circuits. This interruption of power was caused by a brief loss of two of the three PBAPS off-site power sources (EIIS: FK).

Investigation determined that an electrical grid disturbance caused an approximately 16-second loss of two off-site sources. The disturbance was the result of the failure of off-site grid protective relaying (EIIS: RLY) during a lightning storm approximately 35 miles away from the site. The two sources that lost power were lined up to the two plant emergency transformers (EIIS: XFMR), which feed the eight plant emergency busses (EIIS: BU). This condition resulted in de-energization of the emergency busses. The four Emergency Diesel generators (EDGs) (EIIS: EK) actuated on this loss of power condition. The emergency busses were energized by the EDGs as designed. Normal off-site power supplied by the third off-site source was not affected and continued to provide power to two of the four plant non-emergency 13 kV busses.

The Group I, II, and III Primary Containment Isolations on both units resulted in the closure of the Main Steam Isolation Valves (MSIVs) (EIIS: ISV) and other containment process and ventilation piping isolations. The Standby Gas Treatment (SGT) system (EIIS: BH) actuated as designed on the PCIS isolation. On Unit 3, the 86D Outboard MSIV did not initially close. However, the redundant inboard MSIV closed as designed. The 86D outboard MSIV went closed at approximately 0248 hours.

As a result of the Group I PCIS Main Steam Line Isolation, the Main Steam Safety Relief Valves (SRVs) (EIIS: RV) actuated as designed to perform their over-pressure protection safety function. SRVs on both Units 2 and 3 properly relieved pressure with the exception of the Unit 3 71 D SRV. This SRV did not re-close promptly as designed. The 71D SRV re-closed approximately 15 minutes after its actuation at approximately 400 psig reactor pressure. Also, at approximately 0600 hours, the Unit 3 71G SRV did not open when manually actuated from the Main Control Room during reactor pressure control operations. The 71G SRV did initially open and perform its over-pressure protection function when the event initially occurred and was manually actuated prior to 0600 hours for reactor pressure control.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Description of the Event, cont.

Reactor level control was maintained by using the High Pressure Coolant Injection (HPCI) (EIIS: BJ) and Reactor Core Isolation Cooling (RCIC) (EIIS: BN) systems on both Units 2 and 3 taking suction from the Condensate Storage Tank (CST) (EIIS: KA). These systems were proactively placed into service within approximately 10 minutes of the event by Operations personnel. Automatic initiation of these systems was not required since the Level 2 reactor water level set point was not reached. On Unit 2 at approximately 0200 hours, the 'B' Condenser (EIIS: SD) Hotwell level controller (EIIS: LC) failed resulting in the diversion of a limited portion of CST inventory to the Hotwell. This resulted in the automatic swap-over of Unit 2 HPCI / RCIC suction supply from the CST to the Suppression Pool. Condenser Hotwell level control was changed to the 'A' controller and CST level was returned to normal on Unit 2 by approximately 0235 hours. Unit 2 HPCI / RCIC suction was returned to the CST by 0331 hours.

The Unit 3 'D' Suppression Pool Cooling system (EIIS: BO) was initially placed in service by Operations personnel by approximately 0213 hours. At approximately 0235 hours, while initially placing the Unit 2 'B' Suppression Pool Cooling system in service on Unit 2, the E-2 EDG tripped on low jacket coolant pressure. This resulted in not being able to complete placing the Unit 2 'B' Suppression Cooling system in service. Because the 343SU off-site source was supplying power to an emergency transformer, the Unit 3 emergency bus fed from the E-2 EDG (i.e. E-23 bus) transferred to the 343SU off-site source. However, the Unit 2 emergency bus fed from the E-2 EDG (i.e. E-22 bus) remained de-energized preventing the placement of the 'B' Suppression Pool Cooling system in service. The E-22 bus was subsequently energized by 0315 hours using the 343SU off-site source.

The Unit 2 'A' Suppression Pool Cooling system was placed in service by Operations personnel at approximately 0250 hours.

At approximately 0239 hours, the Operations Shift Manager (Emergency Director) declared an Unusual Event following the trip of the E-2 EDG. The Unusual Event declaration was based on previously having a brief loss of off-site power on two of the three off-site power sources coupled with the E-2 EDG inoperability. This entry was made based on discretionary judgment that the level of safety of the plant was potentially degraded. Although not required by the Emergency Plan, the Technical Support Center (TSC) and Emergency Operations Facility (EOF) were conservatively staffed by 0333 hours and the TSC was activated by 0338 hours. The EOF was activated at 0350 hours.

As a result of high water levels in the Suppression Pool (EIIS: NH) caused by SRV, HPCI, and RCIC steam exhausts, as well as loss of containment area cooling on Unit 2 due to loss of the associated plant non-emergency 13 kV bus, containment pressure on both Units 2 and 3 increased to above 2 psig. Unit 2 containment pressure reached 2 psig by approximately 0350 hours while Unit 3 reached the 2 psig pressure by approximately 0541 hours. Both Unit 2 and 3 reactor pressure were maintained above 450 psig while the containment pressure was above 2 psig. Therefore, there was no low-pressure core cooling system initiation signals that were received. Actions were taken to maximize containment area cooling as appropriate. The maximum Unit 2 containment pressure was approximately 2.9 psig at approximately 0800 hours. The maximum Unit 3 containment pressure was approximately 2.3 psig at approximately 1400 hours.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Description of the Event, cont.

Off-site power was restored to the plant emergency busses and the EDGs were secured by approximately 0820 hours. The Unusual Event classification was terminated at 1046 hours based on successful recovery of the normal off-site power sources to the emergency busses.

Unit 2 Follow-up Actions:

The PCIS Group I isolation (Main Steam Lines) was reset by 0645 hours. The MSIVs were re-opened by 0915 hours and the normal plant heat sink (i.e. Condenser) was restored. Containment pressure was reduced as a result of restoring containment area cooling. Once pressure was below 2 psig, the Unit 2 PCIS Group II/III isolations were reset by approximately 1255 hours. A reduction in the Unit 2 Suppression Pool level was initiated using plant procedures at approximately 1330 hours. The Unit 2 Suppression Pool level reduction was completed by 2130 hours. The scram was reset by 2155 hours. Shutdown Cooling was placed in service at approximately 0200 hours on 9/16/03 and the reactor was in the cold condition by 0425 hours on 9/16/03.

Unit 3 Follow-up Actions:

The PCIS Group I isolation (Main Steam Lines) was reset by 0528 hours on Unit 3 and appropriate MSIVs were re-opened by approximately 1115 hours. As a result of the increased Suppression Pool inventory, a temporary procedure was developed to reduce the containment pressure below 2 psig by lowering the Suppression Pool level. This document was approved by 1600 hours and the Suppression Pool level reduction was complete by 2040 hours. The PCIS Group II / III isolation was reset by approximately 2010 hours and the scram was reset on 9/16/03 at 0105 hours. Shutdown Cooling was placed in service at approximately 0115 hours on 9/16/03 and the reactor was in the cold condition by 0121 hours on 9/16/03.

Reporting of the Event:

The notification of the Unusual Event classification was made by 0254 hours.

In accordance with 10CFR 50.72, prompt NRC notifications were initially completed by approximately 0306 hours on 9/15/03 to report the event including the declaration of the Unusual Event. The Emergency Response Data System (ERDS) was promptly activated and other subsequent event updates were provided to the NRC over the Emergency Notification System.

This report is being submitted pursuant to 10CFR50.73 (a)(2)(iv)(A) due to valid actuations of the Reactor Protection, Primary Containment Isolation, SGT, HPCI, RCIC, and the Suppression Pool cooling systems on both Units 2 and 3. Also this report is being submitted due to the automatic start of the four EDGs, which are common to both Units 2 and 3.

This report is also being submitted pursuant to 10CFR50.73 (a)(2)(ii)(A) to report a condition on Unit 3 where a Safety Relief Valve did not re-close in a timely manner. This allowed for a faster Reactor Coolant System pressure reduction than what otherwise would have been planned.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Description of the Event, cont.

This report is also being submitted pursuant to 10CFR50.73 (a)(2)(v)(D) to report a condition on Units 2 and 3 where the off-site sources were unavailable to the emergency busses.

Analysis of the Event

There were no actual significant safety consequences as a result of this event. There were no abnormal radioactive releases involved with this event.

All control rods inserted on the reactor scram signal. The Group I / II / III PCIS isolations resulted in the primary containment isolation safety function being met. All isolation valves closed as required except for the Unit 3 86D Outboard Main Steam Isolation Valve (OBMSIV). The redundant inboard valve closed as designed. The 86D OBMISV went closed by 0248 hours.

HPCI, RCIC, RPS, Suppression Pool Cooling and Recirculation Pump Trip safety functions operated as designed. The EDGs initially started and were loaded appropriately. The E-2 EDG tripped at approximately 0235 hours, however, the remaining EDGs were sufficient to provide power to plant safety systems.

There are three PBAPS off-site power sources (i.e. 2SU, 3SU and 343SU). Any two of these three off-site sources have the capability to be tied to the 2 emergency transformers at the site. The two emergency transformers normally power the four Unit 2 emergency busses and the four Unit 3 emergency busses. At the time of the event, the 2SU and 343SU were tied to the two site emergency transformers. A brief loss of power to the 2SU and 343SU resulted in the EDGs actuating on a loss of power signal from the emergency buses. During this event, the 3SU off-site power source was unaffected. This source continued to provide power to the station's #1 and #4 non-emergency busses. The 3SU off-site power source had the capability of being aligned to the plant emergency busses if necessary. Since the other two off-site power sources (2SU and 343SU) were available shortly after the initial event, actions were taken to restore the emergency busses to the 2SU and 343SU startup sources. The 343SU off-site source was available approximately 16 seconds after the initial event to provide power to one of the two site emergency transformers. The 2SU off-site source circuit breaker (SU-25) was closed at approximately 0600 hours to provide power to the other emergency transformer.

Because the 343SU off-site source had been promptly restored to one of the emergency transformers, the tripping of the E-2 EDG at approximately 0235 hours resulted in loss of only one of the two emergency busses fed from this EDG. An analysis has determined that the EDG was possibly inoperable since the last 2-hour test run of the EDG on 9/2/03. This event is bounded by the Updated Final Safety Analysis Report (UFSAR) analysis for loss of off-site power. The design event assumes one EDG does not function and therefore, the 2 emergency busses that are supplied by that EDG are assumed to not be energized. For the event on 9/15/03, however, one of the two emergency busses fed by the E-2 EDG was transferred to the emergency transformer supplied by the 343SU off-site source.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Analysis of the Event, cont.

The scram and PCIS Group I (Main Steam Line Isolation) is bounded by the design basis event entitled, 'Electrical Load Rejection (or Turbine Trip) without Bypass'. During this event, the plant safety systems responded as necessary. This event did not involve operations that exceeded the design basis.

This event is not considered as a Station Blackout (SBO) event since the EDGs started and energized the 4kV emergency buses as designed and the third off-site source was not lost.

An evaluation was performed concerning the independence of off-site sources that feed the PBAPS site. The off-site power source independence design is in accordance with committed NRC design criteria.

On Unit 3, the SRVs operated as necessary to provide over-pressure protection for the reactor vessel as a result of MSIV closure due to the Group I PCIS isolation. Therefore, the over-pressure protection safety function was satisfied. The 71G SRV properly functioned to provide over-pressure protection and was used for pressure control during the event. However, at approximately 0600 hours, the SRV could not be re-opened. This was not significant for reactor pressure control since other SRVs were available to perform this function.

On Unit 3, the issue involving SRV 71D remaining open for about 15 minutes resulted in a larger pressure / temperature reduction than what would normally be desirable. However, this open SRV is bounded by the design basis event entitled, 'Inadvertent Opening of a Relief or Safety Valve'. The SRV closed at approximately 400 psig reactor pressure and was not needed for subsequent plant pressure control evolutions. It was determined that there were no detrimental effects to the reactor coolant system as a result of this event. The reactor coolant system was considered acceptable for continued operations.

Concerning the declaration of the Unusual Event, the entry was made based on the Shift Manager's (Emergency Director) discretion. This entry was made based on discretionary judgment that the level of safety of the plant is potentially degraded. The entry conditions were not met for the Unusual Event classification for 'Loss of all Offsite AC Power for Greater than 15 minutes to Essential Busses'.

An engineering evaluation was performed concerning the pressure response of the Units 2 and 3 containments. It was determined that the rise in containment pressure was an appropriate response for this type of event. A significant amount of inventory was directed to the Suppression Pool due to the PCIS Group I isolation (SRV exhaust to the Suppression Pool) and the use of HPCI and RCIC for level / pressure control. The rise of the Suppression Pool level resulted in the compression of the atmosphere in the Suppression Pool. The higher pressure resulted in the Drywell to Suppression Pool Vacuum Breakers opening. The opening pressure of the vacuum breakers is nominally 0.5 psid. Therefore, the rising water level in the Suppression Pool raised pressure in the Suppression Pool that resulted in raising the pressure in the Drywell. There were no leaks from the Reactor Coolant System that contributed to the rise in containment pressure.

Analysis of the Event, cont.

The Unit 2 Suppression Pool level was reduced using normal plant procedures that are used at containment pressures below 2 psig (i.e. Group II PCIS isolation reset). The Unit 3 Suppression Pool level was reduced using a temporary procedure that allowed for opening of appropriate containment isolation valves to drain the Suppression Pool while the containment pressure was above 2 psig. The risk of opening these isolation valves while the PCIS Group II signal was still present was determined to be insignificant since the drain line is a small bore pipe, the valves could be expeditiously closed if required and there was no actual event involving the potential release of radioactive material. Technical Specification actions for high Suppression Pool water level conditions were complied with.

Licensed operator performance in response to the dual unit scram was reviewed. It was concluded that operations response was very good. There were no significant human performance issues involved with the event.

A Conditional Core Damage Probability (CCDP) study was performed. The results of this analysis determined that this event had minor risk significance.

Cause of the Event

The electrical grid disturbance that affected the PBAPS site was the result of less than adequate protective relay performance associated with high voltage transmission lines located approximately 35 miles away from the site. It has been determined that primary and back-up protective relaying were disabled by a mechanically failed fuse on the primary and loose electrical connection on the backup. Other contributing causes involving design and maintenance on other protective relaying were also noted. A formal root cause evaluation is in progress. Underlying causes to the failures include less than adequate preventive maintenance and testing of the associated protective relaying equipment.

The E2 EDG trip was caused by low jacket water pressure approximately one hour into the event. The low jacket water coolant pressure has been attributed to combustion gas entering the jacket water system through leaking copper gasket(s) at the cylinder liner adapter seals. A formal root cause investigation is in-progress with focus on inadequate initial adapter gasket pre-load in combination with stress relaxation of the gasket over time.

The failure of the 71D SRV to re-close once actuated is being thoroughly investigated in accordance with the site Corrective Action Program (CAP). The SRV was disassembled and inspected to determine the cause of the SRV not re-closing. It was determined that the pilot valve in the SRV did not re-seat properly and therefore, the SRV remained open. A failure analysis laboratory determined that tightly adhered foreign material on the pilot valve disc might have caused the pilot valve disc from properly re-closing.

The failure of the 71G SRV to be subsequently opened was due to degradation of the air operator diaphragm for the SRV. The degradation was due to accelerated aging caused by exposure to high temperatures. The high temperature condition was apparently caused by leaking packing material that isolates the air actuator from the second stage steam space. Further cause evaluation analyses are in-progress in accordance with the site Corrective Action Program.

Cause of the Event, cont.

The failure of the 86D MSIV to close is being thoroughly investigated in accordance with the site Corrective Action Program (CAP). The valve was disassembled and thoroughly evaluated. It was determined that there were no in-body concerns with the valve and that the most likely failure cause was external to the valve (i.e. actuator or actuator sub-components such as solenoid valves, manifold, etc).

Corrective Actions

The protective relaying associated with the off-site power sources was repaired. Other design, maintenance and testing enhancements are being pursued to upgrade the reliability of the electrical grid protective relaying in proximity to the PBAPS station.

Repairs were made to the E-2 EDG to repair the combustion gas leakage into the jacket water cooling system. Extensive testing and analysis has been performed on all four EDGs at PBAPS. Enhancements have been made to the monitoring program for EDG performance. Improvements will be made to the EDG maintenance practices with regards to the installation of cylinder liner adapter seals. A formal root cause evaluation is in progress and other appropriate corrective actions will be performed in accordance with the Corrective Action Program.

The Unit 3 71D and 71G SRVs were removed and replaced with factory refurbished SRVs. Other SRVs on Unit 3 were also refurbished. An extent of condition review for other SRVs on both Units 2 and 3 was performed. It was determined that the PBAPS SRVs currently installed are highly reliable.

The actuating control components of the 86D MSIV were replaced. An extent of condition review for other MSIVs on Units 2 and 3 was completed.

Previous Similar Occurrences

There were no previous events identified involving a Peach Bottom dual unit scram initiated by an off-site grid disturbance issue.

NRC FORM 366B U.S. NUCLEAR REGULATORY COMMISSION
(1-2001)

LICENSEE EVENT REPORT (LER)

FAILURE CONTINUATION

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COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	SB	RV	T020	Y					
D	SB	RV	T020	Y					
B	SD	LC	F130	N					